TECHNICAL NOTE
ORP/TAD-77-1

Evaluation of Tritium Recycle and Buildup in a Pressurized Water Reactor

THE UNITED STATES
ENVIRONMENTAL PROTECTION AGENCY
Office of Radiation Programs

March 1977

TECHNICAL NOTE
ORP/TAD-77-1

Evaluation of Tritium Recycle and Buildup in a Pressurized Water Reactor

C. Bruce Smith

U.S. Environmental Protection Agency
Office of Radiation Programs
Technology Assessment Division
Energy Systems Analysis Branch

March 1977

PREFACE

The Office of Radiation Programs (ORP) of the Environmental Protection Agency carries out a national program designed to evaluate public health impact from ionizing and nonionizing radiation, and to promote development of control necessary to protect the public health and the environment. This report provides the technical information necessary for ORP to evaluate the environmental aspects concerning tritium recycle in a pressurized water reactor (PWR). This information is important since the impact of tritium recycle is significantly affected by the design and operation of PWRs. Results of this paper may be used as input into dose models to analyze the total environmental impacts of tritium released for various tritium recycle operations.

David S. Smith
Director
Technology Assessment Division (AW-459)

TABLE OF CONTENTS

TABLES

FIGURES

SUMMARY

Recycling radioactive liquid wastes at light-water cooled nuclear power reactors is becoming a popular method of limiting release levels to as low as practicable. As a result of recycling liquid waste, almost all of the radioactive materials in a liquid waste stream can be removed by filters, demineralizers, reverse osmosis units, or evaporators. However, tritium in the liquid waste streams cannot be selectively removed by the aforementioned conventional liquid radwaste treatment equipment. Combined with its relatively long half life (12.33 years), recycling results in a buildup of large volumes of tritium contaminated liquids.

In a boiling water reactor (BWR), the production of tritium is primarily in the fuel and control rods. The zirconium cladding around these rods prevents most of the tritium from escaping from the rods to the reactor liquids. Thus, the buildup of tritium in a BWR is not considered a significant plant problem. In a pressurized water reactor (PWR), tritium is also produced in the fuel and control rods, as well as by production from chemicals (boron, lithium) in the primary coolant liquids. Since the quantities of tritium introduced into the primary coolant system of a PWR are much higher than in a BWR, the management and impact of tritium at a PWR must be evaluated. It should also be noted that most of the tritium produced in the fuel of PWRs or BWRs remains in the clad fuel and control rods and is eventually transported to the reprocessing plant.

To evaluate the magnitude of the potential tritium handling problem at PWRs and to develop recommendations for solutions, a computerized model of tritium buildup in a PWR was developed. The results from this model will be used to assess the potential environmental impact of tritium recycle for the entire nuclear power reactor industry and will be instrumental in developing standards or guidance concerning tritium recycle. The model incorporates six compartments - primary coolant system, liquid radwaste system, condensate storage tank, spent fuel pool, steam generators, and the remainder of the secondary system; and 4 release pathways - containment building leakage, primary-to-secondary leakage, evaporation from the spent fuel pool and refueling canal and intentional removal from the primary system to limit the buildup of tritium in the coolant. A reactor operator can control the buildup of tritium in the primary coolant system of a PWR by utilizing several control options - the intentional removal of liquids from the primary coolant system with subsequent offsite disposal, the intentional environmental release of liquids from the liquid radioactive waste system, and an increase of the evaporation rates from the spent fuel pit and the refueling water canal. To illustrate the effects of tritium recycle, a total recycle system with intentional removal of

primary coolant liquids was selected as the operator control mechanism. Thus, the planned release of tritium to the environment from the liquid radioactive waste system was not included in this analysis. EPA has not endorsed any of the previously mentioned control mechanisms, and will reserve judgment until a complete evaluation has been made. The computer model was used to analyze the environmental tritium releases, buildup of tritium in the reactor, volumes and activities of tritiated water that must be removed to control tritium buildup, the effects of various source terms on tritium buildup, and the sensitivity of the buildup to various secondary system parameters. Generally, the following conclusions were reached for the base case plant (discussed later in this report) that is presented:

(1) Based on the reactor system analyzed in this paper, intentional removal of tritium from the primary coolant system becomes necessary during the 40 year life of the reactor if the production and release of tritium in the primary coolant system is higher than 625 curies per year. In the remainder of this paper this production and release is termed the "source" term. For source terms greater than 1,000 curies/year, removal becomes necessary within 5 years of initial operation of the reactor. The removal (termed intentional removal) involves either the removal of volumes of tritiated primary coolant system liquids for solidification and offsite disposal or selective removal (and disposal) of tritium from the primary coolant liquids. This removal is necessary to limit the concentration of tritium in the primary coolant to established operating conditions for the reactor (assumed in this paper to be 2.5 μCi/ml of H^3).

(2) For a source term of 700 curies/year, about 80% of the losses from the reactor during the reactor life are equally distributed between radioactive decay and uncontrollable releases to the environment. For larger source terms, intentional removal becomes the dominating plant loss mechanism. This removal is necessary for the reasons indicated in (1) above.

(3) During the 40 years of operation of a reactor whose source term is 700 curies per year, about 80% of the uncontrollable environmental releases are equally distributed between evaporation and leakage to the containment building with subsequent release to the environment.

(4) The rate of steam generator blowdown in the secondary system of a PWR has little effect upon losses or buildup of tritium at a reactor. However, if there is no primary-to-secondary leakage, there is a significant reduction in environmental losses from the reactor in conjunction with an increase in the volumes of tritiated liquids that must be intentionally removed from the primary coolant system.

In general, it is concluded that for the base case presented and a source term production of 700 curies per year that (1) uncontrolled environmental releases can be reduced during the 40 year life of the reactor by 65% or more by using total recycle of tritiated liquids*; (2) several thousand curies and several hundred thousand gallons of tritiated water (with a tritium concentration of approximately 2.5 μCi/ml) must be intentionally removed for disposal or treatment over the plant life if "total" tritiated liquid recycle is undertaken; and (3) about one million gallons of water and 5,000 curies of tritium will be remaining in the reactor systems at the end of its planned 40 year life.

*NOTE Removal of tritium from the reactor is necessary to maintain the primary coolant concentration of tritium to the established operating conditions assumed in this paper (2.5μCi/ml). The removed tritium is also assumed not to be available as an environmental contaminant.

EVALUATION OF TRITIUM RECYCLE AND BUILDUP IN A PWR

I. INTRODUCTION

The major modes of production of tritium in a PWR is by fissioning in fuel rods, neutron capture reactions with poison material used in control rods, and activation of primary coolant additives, such as boron and lithium. The tritium produced in the fuel rods and control rods is released to the primary coolant either through defects in the rod claddings or by diffusion through the cladding. Tritium eventually contaminates the liquid radwaste system, the secondary coolant system, the spent fuel pit, and the condensate storage tank either during routine operations or refueling operations. To model the buildup and release of tritium at a PWR employing recycle of tritium contaminated liquids, the interrelationships among systems and the movement of tritium to and from these systems must be analyzed. The following sections describe the physical and mathematical models of tritium recycle and the results of selected parametric studies.

II. PHYSICAL MODEL OF ACTIVITY TRANSPORT

The relationships of the PWR systems, or compartments, are illustrated in a schematic model in Figure 1. The PWR systems containing tritium can be separated into the following functional compartments: (1) primary system; (2) liquid radwaste system and reactor makeup water storage tank (RMWST); (3) spent fuel pit; (4) refueling water storage tank; (5) steam generators; and (6) the remainder of the secondary system. A block diagram of the transport of tritium into and out of these compartments is presented in Figure 2.

Actually there are more leakage pathways and routing options available in a PWR than are presented in Figures 1 and 2. However, in modeling the movement of tritium in a PWR some pathways may be combined or even eliminated without significantly affecting the overall modeling. For example, the leakage of liquids from the primary coolant system into the containment building (L) includes other leakage pathways. Some of these leakages are from the auxiliary building, radwaste building, spent fuel pit, and from various tanks which are collected and routed back to the liquid radwaste system or which contribute as minor gaseous release pathways. These leakages can be accounted for by either adjusting parameters, such as the shimbleed rate (which may also be routed from the primary system ultimately to the radwaste system), or by increasing the leakage to the containment building. Also, the components of the liquid radwaste system may be modeled by using only the reactor makeup water storage tank (RMWST) volume, which herein is considered to include the liquid radwaste system volumes and the actual RMWST volume. This assumption is realistic since liquid radwaste system components do not selectively remove tritium from the liquid stream. Another option is the method of indicating the percentage of tritium recycle employed. This percentage can be expressed either as the fraction of shimbleed flow, primary coolant leakage, makeup flow from the RMWST to the primary coolant system, or any combination of the three. It should also be noted that the computer program used for the parametric analysis of tritium buildup includes a provision for limiting the primary coolant concentration to a predetermined level by increasing the intentional removal (R) (see table 2), as necessary. R simply indicates some type of planned or intentional removal of primary coolant liquids from the primary coolant system (or other systems with high tritium concentrations) in order to maintain the tritium concentration within the technical specifications of the reactor. This removal may be some type of selective removal of tritium from the liquids or removal of large volumes of tritiated liquids for disposal.

The limiting primary coolant tritium concentration that was chosen was 2.5μCi/ml. This value, or one within several tenths of 1 μCi/ml, was selected because of concerns with potential occupational exposures in the reactor building atmosphere and spent fuel pit areas during operation and refueling[1,2,3]. Based on assumed values of primary coolant leakage rates into the closed containment building, this concentration limits the buildup of tritium in the containment building atmosphere to levels previously indicated to be acceptable by utilities. Many of the recent PWR's are being designed with a continuous purge of the containment building atmosphere, thus the limiting primary coolant concentration may be increased significantly (by at the least a factor of 2), since the equilibrium containment concentrations of tritium would be substantially lower. For these PWR's, many of the results and conclusions presented in this paper would be altered in a manner which is somewhat proportional to the limiting coolant concentration used. The calculated removals and environmental releases for several higher limiting primary coolant concentrations are presented in table 1 and the results are discussed in section IV.D.

The following sections discuss (1) tritium buildup in each of the compartments during routine operation and shutdown and (2) the relationship between compartments.

A. Routine Operations

1. Compartment 1 - Primary Coolant System

Tritium in the primary coolant system can be transported from the system by leakage into the containment building and the auxiliary building. The tritium that remains in the water phase of these leakages is drained to the liquid radwaste system. The tritium that is in the gaseous phase is eventually released through the containment building ventilation system. A small side flow of primary coolant (shimbleed) may be processed through the liquid radwaste system and collected in the RMWST. Primary coolant makeup is then returned to the primary system from the RMWST. Other losses from the primary system include primary to secondary leakage, evaporation of tritium from the refueling canal during refueling shutdown, radioactive decay, and intentional removal (discussed previously).

2. Compartment 2 - Liquid Radwaste System and RMWST

As indicated previously relative to tritium, the liquid radwaste system can be modeled as simply a tank which includes all the liquid radwaste system tanks and components and the reactor makeup water storage tank (RMWST). The additions to this system come from shimbleed from the primary coolant system, the collection of containment building liquids contaminated by primary coolant leakage, or via routing of steam generator blowdown liquids to this system. Losses from this system can result from controlled discharges to the receiving waters of the plant, makeup to the primary coolant system, radioactive decay, and continuous makeup to the spent fuel pit. As indicated tritium can be transported to the liquid radwaste system by the routing of steam generator blowdown liquids; however, in the base case analyzed in this paper, it is assumed that the blowdown is routed instead to the main condenser. Another option, which is analyzed in section IV. C. 1, is to route the blowdown directly to the environment.

3. Compartment 3 - Spent Fuel Pit

During routine operations tritium is added to the spent fuel pit by makeup from the RMWST. Removal of tritium from the reservoir is either by radioactive decay or evaporation. During refueling shutdown a portion of the spent fuel pit water becomes mixed with the primary coolant system in the refueling canal. As previously indicated, during refueling shutdown tritium is removed from the refueling canal by evaporation and decay.

4. Compartment 4 - Refueling Water Storage Tank

Tritium is transported to the refueling water storage tanks (RWST) (often called condensate storage tanks) mainly as a result of refueling operations. During refueling the refueling canal is flooded with a large volume of water from the RWST. At the end of refueling operations, this water, which is now contaminated with tritium, is returned to the RWST. The only loss from the RWST during routine reactor operation is radioactive decay.

5. Compartment 5 and 6 - Secondary System Steam Generators and the Remaining Secondary System

During routine operation tritium may leak or diffuse from the primary coolant system into the steam generators of the secondary system. From the steam generators a small fraction of the tritium is carried over in the steam to the turbines and returned to the steam generators in the feedwater. If the steam generators are blowndown, tritium can be transported to the environment, the RMWST (via the radwaste system), or the main condenser. Some of the tritium which accumulates in the remainder of the secondary system is released to the environment by steam leakage, steam jet air ejector exhaust discharges, and condensate leakage to the turbine building.

B. Shutdown

During refueling shutdown, a fraction of the primary coolant system is drained to the liquid radwaste system for processing and then retained in the RMWST for the duration of refueling operations. The remaining primary coolant water in the refueling canal is then mixed with a large volume of water from the refueling water storage tank (RWST) and a small fraction of the water from the spent fuel pit. Tritium losses from the reactor during refueling operations are from radioactive decay, evaporation from the spent fuel pit, and evaporation from the refueling canal in the containment building. Figure 3 illustrates the losses and liquid volumes in the compartments during refueling. (Note that the two compartments of the secondary system are not disturbed during a refueling shutdown.)

III. MATHEMATICAL MODEL OF ACTIVITY BUILDUP

Figure 4 illustrates mathematically the transport to and from each of the six compartments previously discussed. The six compartments can be mathematically described by the following differential equations.

1) dA1(t)/dt = P + N*A1(t) + Q*A2(t)
2) dA2(t)/dt = T*A1(t) + V*A2(t) + H*ASG(t)
3) dA3(t)/dt = U*A2(t) + W*A3(t)
4) dA4(t)/dt = -LAMBDA*A4(t)
5) dASG(t)/dt = LL*AS*A1(t) + G*ASG(t) + C*ASEC(t)
6) dASEC(t)/dt = (1-LL)*AS*A1(t) + D*ASG(t) + F*ASEC(t)

The left side of each equation is the time rate of change of activity in that particular compartment. The right side of each equation is composed of additions and removals of tritium for each of the compartments. All terms are in units of Curies/month. The terms of the right side of these equations are:

Equation 1	P	Production of tritium in compartment 1 (primary coolant system) plus release of tritium into the primary system from fuel and control rods
	N*A1(t)	Radioactive decay; intentional removal; primary system leakage to the containment building, auxiliary building, and secondary system; and shimbleed for compartment 1
	Q*A2(t)	Reactor water makeup from compartment 2 (RMWST) and liquid radioactive waste system
Equation 2	T*A1(t)	Addition of tritium to compartment 2 via leakage collected in building drains and primary system shimbleed
	V*A2(t)	Radioactive decay in compartment 2; reactor water makeup to compartment 1 from compartment 2; water makeup to compartment 3 (spent fuel pit)

	H*ASG(t)	Blowdown from compartment ASG (steam generators) if blowdown is routed to compartment 2
Equation 3	U*A2(t)	Water makeup to compartment 3 from compartment 2
	W*A3(t)	Radioactive decay and evaporation from compartment 3
Equation 4	-LAMBDA*A4(t)	Radioactive decay in compartment 4 (refueling water storage tank)
Equation 5	LL*AS*A1(t)	Primary to secondary leakage and diffusion of tritium into the steam generator liquids
	G*ASG(t)	Steam generator blowdown; carryover of tritium by the production of steam in the steam generators; radioactive decay in the steam generators
	C*ASEC(t)	Feedwater to the steam generators from compartment SEC (the remainder of the secondary system)
Equation 6	(I-LL)*AS*A1(t)	Primary to secondary leakage and diffusion of tritium to the steam phase of the steam generators
	D*ASG(t)	Steam generator carryover and blowdown of tritium from the steam generators to the remainder of the secondary system
	F*ASEC(t)	Radioactive decay; steam leakage; condensate leakage; air ejector partitioning; steam generator feedwater from the remainder of the secondary system

The definitions of the components of each term are presented in Table 2. Each of these equations has been solved for the activity in each compartment as a function of time*. The tritium releases from each compartment can be calculated by integrating the activities to obtain an average activity during an interval of time. The average activities are then multiplied by the appropriate removal fractions to obtain tritium releases for the time interval. Description of the computerized models of the six compartments is not presented in this paper.

IV. PARAMETER SELECTION

The computer program that utilizes the equations in section III requires the input of 33 parameters. The values of the parameters selected for the base case reactor system are presented in table 3. For the computer analysis, these parameters are held constant throughout the 40 years of operation of a reactor. Naturally, for an actual reactor system many of these parameters will vary considerably from year-to-year or even month-to-month. Thus, the constant values chosen for the parameters are considered the expected long-term average over the life of the reactor. The following paragraphs indicate the basis of these values, and briefly the significance of some of them. The symbols and values of the parameters correspond to the listing presented in table 3.

A. Trojan FSAR Data, Chapter 11

The following parameter values were selected from the Trojan Nuclear Plant FSAR[5]: V1,V2,V3,V4,V1REF,LIMIT,FSFP,ELSPO,ELSFPR, and ELRCR. For other PWRs the four volumes selected (V1,V2,V3,V4) may vary considerably; however, for almost all large PWRs, the total volume of these systems will be around 1,000,000 gallons. The selection of the volumes of these systems is not significant to the overall results presented in this paper. The fraction of mixing of the spent fuel pit water (FSFP) with refueling water during refueling shutdown for any reactor will be a small fraction of the spent fuel pit water available (V3) and is not a critical parameter. The three evaporation rates, ELSPO, ELRCR, and ELSFPR are extremely critical to the results of this paper and to the tritium recycling operations of a PWR. For a sensitivity analysis of these three parameters to be worthwhile, a better data base is required. It can be expected that the water temperatures of these volumes are representative of operating PWRs. However, since evaporation rates are very sensitive to changes in water temperature, small changes in the water temperatures can significantly affect the evaporation rate of tritium. In actual operation we would not expect the tritium evaporation rates to decrease significantly from the values presented in table 3; however, the rates can be increased by approximately factors of 2-4 by increasing water temperatures 30-40°F. As illustrated in Figure 10 of this paper, evaporation rate increases of a factor of 2-4 would significantly affect environmental releases and intentional removal rates during the lifetime of the reactor.

B. PWR-Gale Code NUREG-0017

The Nuclear Regulatory Commission has issued NUREG-0017[6] which discusses the parameter values used for LPS (primary to secondary leakage), LST (steam leak), LCOND (condensate leak), FCO (Carryover fraction in the steam generator) and LL (indicator that primary to secondary leakage is into the water phase of the steam generator). We assumed that the primary to secondary leak rate includes diffusion of tritium through the steam generator walls. If a large total transfer of tritium to the secondary system is assumed, its effect upon tritium buildup and release at the reactor can be ascertained generally from figure 10 of this paper. Variations in the parameter are of less consequence than variations in water evaporation rates that were discussed in section IV.A. We also assumed that the carryover of tritium in a steam generator (FCO) is the same as the iodine carryover presented in NUREG-0017. The moisture carryover of PWR steam generators is much lower (about 0.1%) and may be more representative of tritium carryover. This parameter requires some verification; however, this parameter (FCO) has very little effect upon tritium buildup or release at a PWR.

C. Other Parameters

A number of parameter values in table 3 indicate the options chosen for operating the reactor. First it was decided not to include planned releases of tritium from the liquid radioactive waste system, i.e., this reactor is considered as a total tritium recycle plant. Thus FRM, FRC and FS are 1.0. The parameter values for FBV, BLDN, INDBN, and RR specify the mode of operation of steam generator blowdown. Variations of steam generator blowdown options are discussed later in this paper. The secondary system flow rates and liquid masses (STR,MS,MSG) were selected based on reasonable values from currently designed large PWRs. Variations in these parameters do not have a significant effect on the results of this paper. One of the secondary system parameters, the tritium partition factor for the air ejector (PAG), was estimated from available information in PWR safety analysis reports; however, any reasonable value chosen should not significantly affect the results of this paper. The years of operation (YROP), the range indicator (RANGE), and the volume of refueling water used during refueling (V4REF) were chosen arbitrarily. If YROP is chosen to be less than 40 years, results of interest can be extracted from the tables and figures presented. It is assumed that almost all the refueling water available (V4REF) would be used during a refueling shut down. The shimbleed (S) rate that was chosen is a reasonable value for large PWRs. It could vary by a factor of two or more; however, its value does not significantly affect the results of this paper. The containment building leak rate (L) and fraction of this leakage that remains in the liquid phase (FL) (and thus, is returned to the liquid radioactive waste system) were chosen somewhat arbitrarily. Values of FL were not available; however, in choosing the value of 0.5 it is assumed that about half of the leakage will be hot (thus, being in a vapor phase) and about half will be relatively cold (thus, remaining as a liquid). The value selected for L is higher than the value presented in NUREG-0017. The primary coolant leakages from systems in the containment building and auxiliary buildings were assumed to be included. Any leakage remaining as a liquid was assumed to be eventually returned to the liquid radioactive waste system. More information concerning these two parameters (L and FL) is necessary, since they can affect the results of this paper significantly.

D. Conclusions of Parameter Values

The major conclusions concerning parameter selection are that more information is necessary for the parameter values of (1) evaporation rates (ELSPO, ELSFPR, ELRCR), (2) the transfer of tritium to the secondary system (LPS), and (3) the primary system leakage to the containment and auxiliary building (L) and the fraction of this leakage remaining in the liquid phase (FL).

The values of these parameters significantly affect the results of this paper. The values chosen for this paper are reasonable, and the results of this paper may be generally applied to a large PWR. However, for a tritium recycle analysis of a specific PWR, these parameter values should be derived for the specific system of that PWR.

E. Ranges of Parameter

The expected ranges for the parameters in table 3 are:

1)	V1+V2+V3+V4	800,000 - 1,250,000 gallons
2)	V1REF	30,000 - 40,000 gallons
3)	V4REF	200,000 - 350,000 gallons
4)	YROP	30 - 40 years
5)	L	UNKNOWN
6)	LIMIT	1.0 - 3.0μCi/ml applicable NRC Regulatory Guide for the majority of PWRs. 5.0 -7.5μCi/ml is applicable for continuous purge systems in containment buildings.
7)	S	0.7 - 3.0 gpm

8)	FSFP	0.10 - .20
9)	ELSPO	25 - 150 gallons/day (estimate)
10)	ELSFPR	75 - 300 gallons/day (estimate)
11)	ELRCR	300 - 1500 gallons/day (estimate)
12)	LPS	0 - 50 gallons/day (estimate of leakage rate which also compensates for diffusion)
13)	FL	0.5 - 1.0
14)	BLDN	0 - 350,000 lb/hr
15)	LST	340 - 1700 lb/hr
16)	LCOND	UNKNOWN
17)	MS	1,500,000 - 5,000,000 lb
18)	MSG	400,000 - 500,000 lb. total for all steam generators
19)	FBV	0 - .33
20)	PAJ	UNKNOWN
21)	STR	1,300,000 - 1,700,000 lb/hr
22)	FCO	0.0008 - 0.002 or (moisture carryover) 0.01 - 0.10 (iodine carryover)

V. ANALYSIS AND RESULTS

The Trojan reactor components and operational characteristics were chosen as the base case for a parametric analysis of the buildup and release of tritium at a large PWR. Table 3 presents the values of the parameters (which are defined in table 2) for the base case. Each analysis is based on the values presented in table 3 unless a different parameter is specifically indicated in the discussion of the analysis. The parametric study can be divided into three groups which are discussed in the following sections.

A. Annual Tritium Source Term

Four annual tritium source terms (350; 700; 1,000; 1,400 curies) were arbitrarily chosen for analysis using the base case parameters presented in table 3. Figures 5 and 6 indicate the tritium buildup in the primary coolant system and the total reactor plant over a 40 year time period. In figure 5 the primary coolant activities represented by curves 2, 3, and 4 reach their limiting concentration (which corresponds to about 2.5 μCi/ml concentration) within approximately 12 years. This limiting concentration (about 850 curies in the primary coolant system for the base case) for the primary coolant system is controlled by the intentional removal of primary coolant liquids. These removed liquids can either be solidified and disposed of at a waste burial ground or treated for selective removal of tritium, if a system is available, and the tritium then disposed of as a solid. The increase in tritium activity in the primary coolant system is a slow process and is also greatly influenced by the activity levels in compartment 2 (RMWST and liquid radioactive waste system). Thus, intentional removal of tritium can be accomplished effectively from either the primary system, the RMWST, or the liquid radioactive waste system on a continuous or batch basis. This flexibility allows selective removal techniques to be utilized that have low flow rates but which can operate continuously for long periods of time. Curve 1 indicates that a production term of 350 Ci/yr would not result in the limiting concentration during the normal lifetime of a reactor. Figure 7 indicates the tritium removal rates (in curies or gallons of tritiated liquids) that will be required for various tritium source terms. Removal becomes necessary when the annual tritium source term in the primary coolant system reaches 625 curies/year. Also for each incremental increase (above 625 curies) of 100 curies in the annual source term , the total volume of tritiated liquids that must be removed (over the lifetime of the reactor) from the primary coolant system increases approximately 400,000 gallons.

As indicated in figure 8, during the first few years of reactor operation, it is not necessary to intentionally remove tritium from the primary coolant system. However, it should be noted that for source terms larger than about 1,000 curies per year, removal does become necessary within five years after initial operation of the reactor.

B. <u>Analysis of a Standard Annual Tritium Source Term of 700 Ci/year</u>

For an analysis of various aspects of tritium in a PWR, a constant annual source term of 700 Ci/yr was chosen. The selection of this source term was somewhat arbitrary but appears reasonable for the following reasons:

1) 700 Ci/year is consistent with tritium source terms of operating reactors based on available data, as indicated in table 4.

2) 700 Ci/year is a high enough source term so that removal from the primary coolant system may be included in the analysis.

The activity buildup in each compartment and in the total reactor plant is presented in figure 9. It should be noted that the primary coolant system, secondary system, and reactor makeup water storage tank (RMWST) reach a stable activity level much sooner than the spent fuel pit or the refueling water storage tank. The secondary system retains very little tritium and is not a concern at the end of the plant's useful lifetime. The equilibrium level of tritium of this reactor case is approximately 5,400 - 5,500 curies.

Figure 10 presents the tritium losses and retention for the base plant design for a 700 Ci/yr source term. The losses are divided into three groups - radioactive decay, uncontrolled environmental losses, and intentional removal. Decay and environmental losses combined constitute more than 80% of the total tritium losses from the plant. Each of these loss mechanisms are approximately equal in magnitude. Note, the intentional removal could be discharged directly to the environment thus significantly increasing the fraction of liquids discharged. However, in this paper it is assumed that the liquids that are intentionally removed from the primary coolant system are not discharged to the environment.

The environmental losses are divided into three major categories - evaporation (during normal operation from the spent fuel pit and during shutdown from the spent fuel pit and the refueling canal), leakage from the containment building, and secondary system losses (primarily from steam leakage and condensate leakage). About 80% of the environmental losses are from evaporation and leakages to the containment building with both pathways being approximately the same over the 40 year life of the plant. It should also be noted that intentional removal is assumed to be necessary after 11 years 9 months of operation. The hatched area in figure 10 indicates the annual increase in activity that is retained in the plant. After about 30 years, the plant activity is stabilized. Thus, block 3 in figure 10 is indicative of the losses of the system not only for the 40th year but also for each year from the 30th year through the 40th year.

Figure 11 indicates the fraction of the environmental releases that are gases or liquids. Over the 40 year lifetime of the reactor almost 90% of the environmental tritium release is expected to be in the gaseous phase. All of the tritium which is intentionally removed during the 40 years of operation (2,780 curies) and the tritium retained at the end of the 40 year period (5,460 curies) is in the liquid phase. A determination of the ultimate fate of these liquids is outside the scope of this paper.

C. Effects of Secondary System Parameters

1. Reactor Losses as a Function of Steam Generator Blowdown

Figure 12 indicates the effects of recycling steam generator blowdown liquids versus routing them directly to the environment. The increase in the total environmental releases from the reactor is less than 3% for direct release of blowdown. However, the gaseous releases now decreased to about 80% of the total environmental release for the cases with blowdown discharged to the environment, versus 90% for the case where blowdown is recycled. This results from the liquid releases almost doubling while the gaseous releases are decreased by about 8%. For the case of the blowdown routed to the environment, the rate of blowdown has very little effect on the magnitude of the environmental releases. Thus, blowdown rates of 5 gpm or even several hundred gpm (average annual blowdown rates) will produce similar results as those presented in Figure 12.

2. Reactor Losses as a Function of Primary to Secondary Leak Rate

Figure 13 indicates the environmental losses and intentional removal necessary for a reactor with no primary to secondary leakage and one with an 18.7 gpm leakage rate. The annual source term used was 700 curies. The radioactive decay losses and the retention of tritium in the reactor are similar for both primary-to-secondary leakage cases (see figure 10). Several conclusions may be reached concerning the results of zero primary-to-secondary leakage in comparison to a reactor with 18.7 gpm leakage.

1) There are no liquid environmental releases; the gaseous releases are 9% lower; and total environmental releases are 19% lower.

2) Intentional removal becomes necessary almost 3 years sooner.

3) The number of curies of tritium that must be intentionally removed is 61% higher (4,480 curies versus 2,780 curies) and the volumes of liquids that are required to be removed are 67% higher (500,000 gallons versus 300,000 gallons).

D. Effects of Various Limiting Concentrations

Table 1 indicates the impacts of limiting the tritium concentrations in the primary coolant system of a PWR. Two source terms (700 Ci/yr and 1,400 Ci/yr) were chosen and three limiting primary coolant concentrations (2.5, 5.0, and 7.5 μCi/ml) were analyzed for these source terms. As discussed previously, for the base case parameters chosen, intentional removal is necessary if the source term is greater than 625 Ci/yr and the limiting concentration is about 2.5 μCi/ml. For the two higher limiting concentrations selected, intentional removal would not become necessary until a much higher source term is reached. For these two cases the environmental releases, primary coolant concentration, and total curies built up in the reactor would stabilize primarily as a result of radioactive decay in the plant and uncontrolled environmental losses. The significant environmental losses are from the refueling canal evaporation during refueling, evaporation from the spent fuel pit during refueling and routine operation, primary system leakage into the containment atmosphere with subsequent release to the environment, and release via secondary system leakages.

For the 1,400 Ci/yr source term, intentional removal is necessary for the 5 μCi/ml limit but not for the 7.5 μCi/ml limit. Doubling the source term from 700 Ci/yr to 1,400 Ci/yr and doubling the concentration limit from 2.5μCi/ml to 5.0μCi/ml results in approximately the same quantity of primary coolant that must be intentionally removed (300,000 gallons versus 250,000 gallons); however, the curies of tritium intentionally removed from the primary coolant system increases by almost a factor of two (2,785 curies to 4,525 curies). Also the initial time period after reactor startup for commencing removal is fairly close (12 years 9 months versus 13 years 10 months).

A major conclusion from this overall analysis is that as the limiting primary coolant concentration is increased there is a nearly proportional decrease in intentional removal required, an increase in uncontrolled environmental releases, and an increase in the buildup of tritium in the reactor.

VI. SIGNIFICANCE OF RESULTS

The results presented in Section IV will provide the information to make an assessment of the environmental and inplant impacts of decisions related to the recycle of tritiated liquids in pressurized water reactors. The data provided in this paper presents an estimate of the reduction in discharges of tritium to the environment under various conditions; the rate of buildup and the equilibrium level of tritium in the reactor plant liquids; and the amount of tritium which will be discharged to the environment through plant ventilation systems and from the secondary system leakages which may be considered unavoidable.

Further, this paper indicates the time at which the limiting conditions of operation will be attained and the amount of tritium which would have to be intentionally removed in order to meet the coolant specification limits for a given tritium source term. This information will enable an assessment to be made of the rate at which tritium must be removed from the coolant system and subsequently the capacity of the tritium removal systems which must be employed to achieve the desired ends. Because of the slow buildup of tritium in the coolant system, the tritium may be controlled on a periodic, batch basis during plant shutdowns. This will be particularly advantageous since, if it is necessary to add a tritium control system, it will be possible to provide this treatment capability independent of plant operation. Thus, the tritium control system will not impact on the plant reliability and the plant safety during operations.

In summary this paper, while it has not reached conclusions as to the desirability of recycle of tritiated liquids nor the necessity for provision of a tritium control system in a pressurized water reactor, it has documented a realistic estimate of the inplant and environmental release source terms associated with tritium recycle. From this information the impact on plant personnel, plant operation, and the population doses may be directly evaluated, and decisions may be made regarding tritium control options.

REFERENCES

1. Wilson, Robert L. and Jay Y. Lee. "Radioactive Liquid Waste System for a Dry Site," for presentation at the ANS Annual Meeting, Las Vegas, Nevada, June 19-22 (1972).

2. "Technical Paper Oconee Radiochemistry Survey Program," Summary of four presentations presented to the American Nuclear Society, New Orleans, Louisiana, June 8-13 (1975).

3. Final Safety Analysis Report for Sequoyah Nuclear Power Plant, Units 1 and 2, NRC Docket Nos. 50-387 and 388, Appendix 11A.

4. Korn, Granino A. Ph.D., and Theresa M. Korn, Mathematical Handbook for Scientists and Engineers, Page 416, Sylvestor's Theorem, Second edition, McGraw Hill, 1968 (Library of Congress Catalog Card Number 67-16304).

5. Final Safety Analysis Report of the Trojan Nuclear Power Plant, Docket No. 50-344.

6. Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors (PWR-Gale Code), NUREG-0017, Office of Standards Development, U.S. Nuclear Regulatory Commission, April, 1976.

7. Gruhlke, James M., "Pressurized Water Reactor Effluent Discharge Trends in the United States," U.S. Environmental Protection Agency, Office of Radiation Programs, Technology Assessment Division, Energy Systems Analysis Branch, Washington, D.C., Presented at the ANS Annual Meeting, June 13-18, 1976, in Toronto, Canada.

Table 1. Effects of Various Limiting Primary Coolant Concentrations

Primary Coolant Limit (μCi/ml)	Source Term (Ci/yr)	Tritium Buildup After 40 years of Operation (Curies)	Intentional Removal		After 40 Years of Operation (Curies/year)		
			Gallons for 40 years	Curies for 40 years	Intentional Removal	Environmental Releases	Radioactive Decay
2.5	700	5,500	300,000	2,785(1)	105	290	305
5.0	700	6,400	0	0	0	335	355
7.5	700	6,400	0	0	0	335	355
2.5	1,400	5,700	2,900,000	28,220(2)	770	310	320
5.0	1,400	11,300	250,000	4,525(3)	140	595	625
7.5	1,400	12,900	0	0	0	670	710

Intentional Removal Started After:
- (1) 12 years 9 months
- (2) 3 years 7 months
- (3) 13 years 10 months

Table 2. Parameter Identification

Parameter	Definition
A1(t),A2(t),A3(t) A4(t),ASG(t),ASEC(t)	Activity of each compartment as a function of time, Ci
P	Production and release of tritium into compartment 1, Ci/month
N	-LAMBDA-30.4*(L+R+S+LPS)/V1, fractional removal per month from compartment 1 by radioactive decay, leakage, planned removal, shimbleed, and primary to secondary leakage, month^{-1}
Q	30.4*(L+R+S+LPS)/V2, fractional addition per month to compartment 1 as a result of makeup from compartment 2 to maintain a proper liquid volume balance in compartment 1, month^{-1}
T	30.4*(L*FL*FRC + S*FS)/V1, fractional addition per month to compartment 2 from compartment 1 via shimbleed and primary coolant leakage returned to RMWST, month^{-1}
V	-LAMBDA-30.4*(L+R+S+LPS)/FRM*V2-30.4*ELSPO/V2, fractional removal per month from compartment 2 by radioactive decay, makeup to compartment 1, and makeup to compartment 3, month^{-1}
U	30.4*ELSPO/V2, fractional makeup per month to compartment 3 from compartment 2, month^{-1}
W	-LAMBDA-30.4*ELSPO/V3, fractional removal per month from compartment 3 be radioactive decay and evaporation, month^{-1}

Table 2. (Continued)

Parameter	Definition
H	30.4*24*INDBDN*BLDN*(1-FBV)/MSG, fractional addition per month to compartment 2 from blowdown of the steam generators, $month^{-1}$
AS	30.4*LPS/V1, fractional addition per month from compartment 1 to the steam generator liquids or secondary system steam via primary to secondary leakage and diffusion, $month^{-1}$
G	-30.4*24*BLDN/MSG-30.4*24*FCO*STR/MSG-LAMBDA, fractional removal per month from the steam generators by blowdown, moisture carryover in the steam, and radioactive decay, $month^{-1}$
C	30.4*24*STR/MS, fractional addition per month to the steam generators from the remainder of the secondary system via feedwater, $month^{-1}$
D	30.4*24*FCO*STR/MSG + 30.4*24*RR*BLDN*(1-FBV)MSG, fractional addition per month from the steam generators to the secondary system by moisture carryover and blowdown routed to the main condensers, $month^{-1}$
F	-LAMBDA-(LST+LCOND+STR/PAJ)*30.4*24/MS -30.4*24*STR/MS, fractional removal per month from the secondary system by radioactive decay, steam leakage, condensate leakage, air ejector partitioning, and feedwater makeup to the steam generators, $month^{-1}$
V1,V2,V3,V4	Volumes of compartments 1 through 4, gal.

Table 2. (Continued)

Parameter	Definition
MSG,MS	Mass of liquids in the steam generators and the remainder of the secondary system, lbs.
L	Primary system leakage to the containment building, gal/day
S	Shimbleed, gal/day
LPS	Primary to secondary leakage and compensation for diffusion through steam generator tubes, gal/day
R	Removal of tritium from the primary coolant system in order to control the primary coolant tritium concentration, gal/day
ELSPO	Evaporative losses from the spent fuel pit, gal/day
FL	Fraction of L that remains in the liquid phase
FS	Fraction recycle of shimbleed
FRM	Fraction recycle of RMWST flow used as makeup to the primary coolant system
FRC	Fraction of L*FL liquids that are recycled
FBV	Fraction of steam generator blowdown vented in the gaseous phase
FCO	Fraction of tritium carried over from the steam generators to the secondary system steam
BLDN	Steam generator blowdown rate, lb/hr

Table 2. (Continued)

Parameter	Definition
INDBDN	Indicator, equals 1 if BLDN is routed to the RMWST else it equals 0
LST	Steam leakage from the secondary system, lb/hr
LCOND	Condensate leakage from the secondary system, lb/hr
LL	Indicator, equals 1 if LPS is into the steam generator liquids, equals 0 if LPS is into the steam phase in the steam generator
PAJ	Partition factor of the steam jet air ejector
STR	Steaming rate of the steam generators, lb/hr
RR	Indicator, equals 1 if BLDN is routed to the main condenser else it equals 0
LAMBDA	Radioactive decay constant, $month^{-1}$

Table 3 Base Case Parameters

90000	V1 VOLUME PRIMARY SYSTEM, GAL
210000	V2 VOLUME WATER RADWASTE AND COND STR, GAL
394000	V3 VOLUME WATER SPENT FUEL PIT, GAL
350000	V4 VOLUME WATER REFUELING STR POOL, GAL
35000	V1REF VOLUME PRIMARY WATER IN CORE AT SHUTDN, GAL
340000	V4REF VOLUME REFUELING WATER USED DURING REFUELING
40	YROP YEARS OF OPERATION
1	IND2 INDICATOR 0=MULTI-RUNS 1=TABLE, 2=PLOT, 3=ANN
68	L PRIMARY SYSTEM LEAKAGE GAL/DAY
0	INDBDN INDICATOR=1IF BLDN ROUTED TO RMWST ELSE=0
2.5	LIMIT TRITIUM CONC LIMIT IN PRIMARY SYS, UCI/ML
0.1	RANGE INDICATOR FOR CONTROLLING H-3 CONC, UCI/ML
2920	S SHIMBLEED FLOW, GAL/DAY
0.15	FSFP FRACTION MIXING SPENT FUEL WATER W/REFUEL H2O
48.4	ELSPO EVAP LOSS FROM SPENT FUEL PIT OPERATING, GAL/DAY
131	ELSFPR EVAP LOSS FROM SPENT FUEL SHUTDN, GAL/DAY
464	ELRCR EVAP LOSS FROM REFUEL CANAL REFUEL, GAL/DAY
18.7	LPS PRIMARY TO SECONDARY LEAKAGE, GAL/DAY
1	FS FRACTION RECYCLE SHIMBLEED
0	FL FRACTION PRIM COOL LEAK IN LIQUID
1	FRM FRACTION RMWST RECYCLED
8400	BLDN BLOWDOWN RATE, LB/HR
1700	LST STEAM LEAK, LB/HR
2400	LCOND CONDENSATE LEAK SEC SYS, LB/HR
1	FRC FRACTION REACTOR DRAIN RECYCLED
2000000	MS MASS SECONDARY SYSTEM, LB
300000	MSG MASS OF ALL STEAM GENERATORS, LB
1	LL INDICATOR=1, LPS INTO STEAM GEN LIQ
0	FBV FRACTION BLDN VENTED AS GAS
200000	PAJ PARTITION FACTOR AIR EJECTOR
17000000	STR STEAMING RATE, LB/HR
0.01	FCO CARRYOVER FRACTION IN STEAM GENERATOR
1	RR INDICATOR=1, ROUTED TO CONDENSER ELSE=0

Table 4. PWR Tritium Releases[7]

Year	Tritium Releases (Ci/year)
1972	1042
1973	715
1974	685

Note:

1) PWRs (zircaloy Clad Fuel) with at least one prior year of commercial operation.

2) All release data is normalized to a 3400 MWth reactor which operates at full power 80% of the year (292 days). (Note that the operating cases in this paper are for a reactor operating at full power for 11 months of the year).

3) Releases per year are assumed to be the same as the source term for the year since none of the reactors employ tritium recycle.

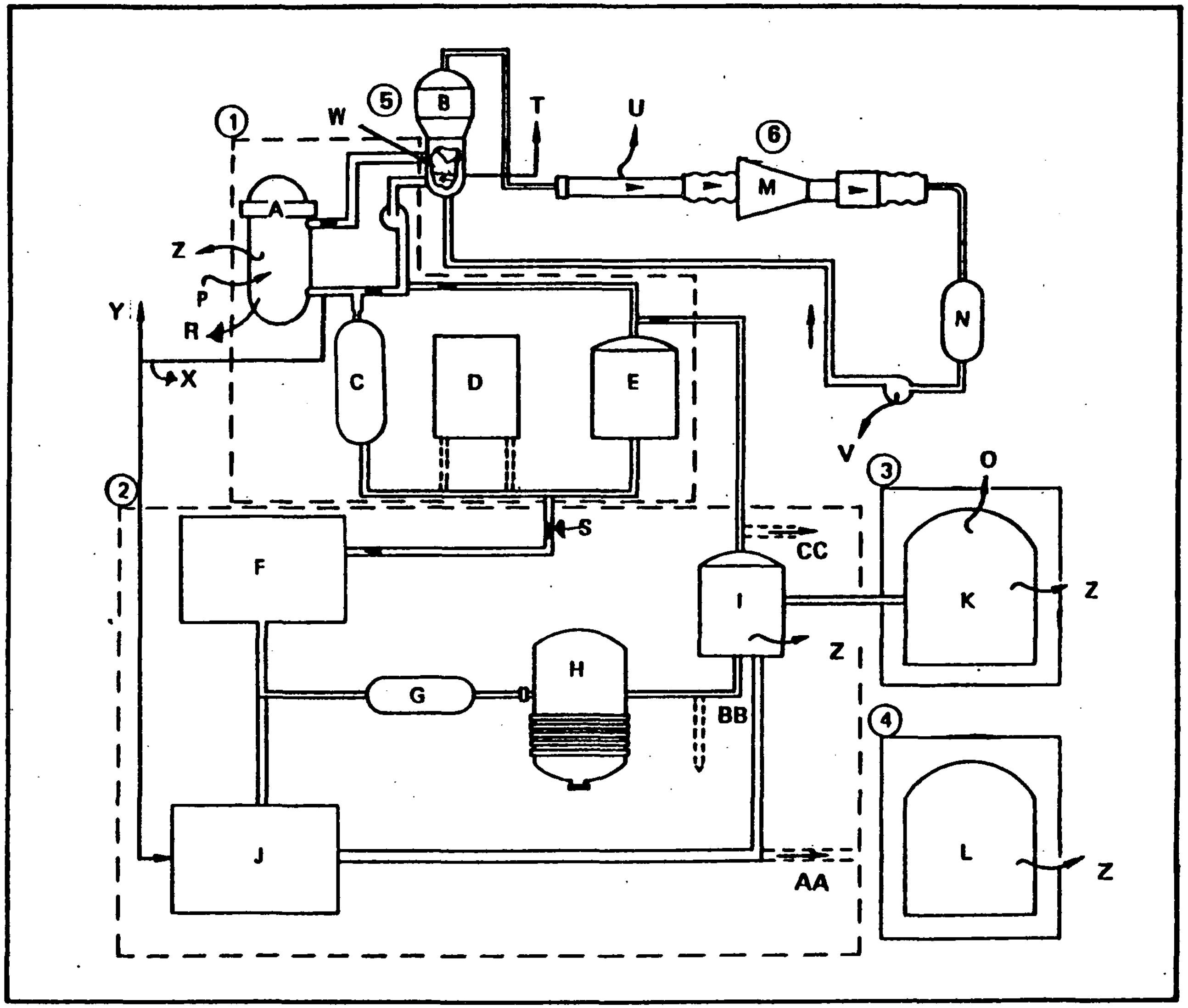
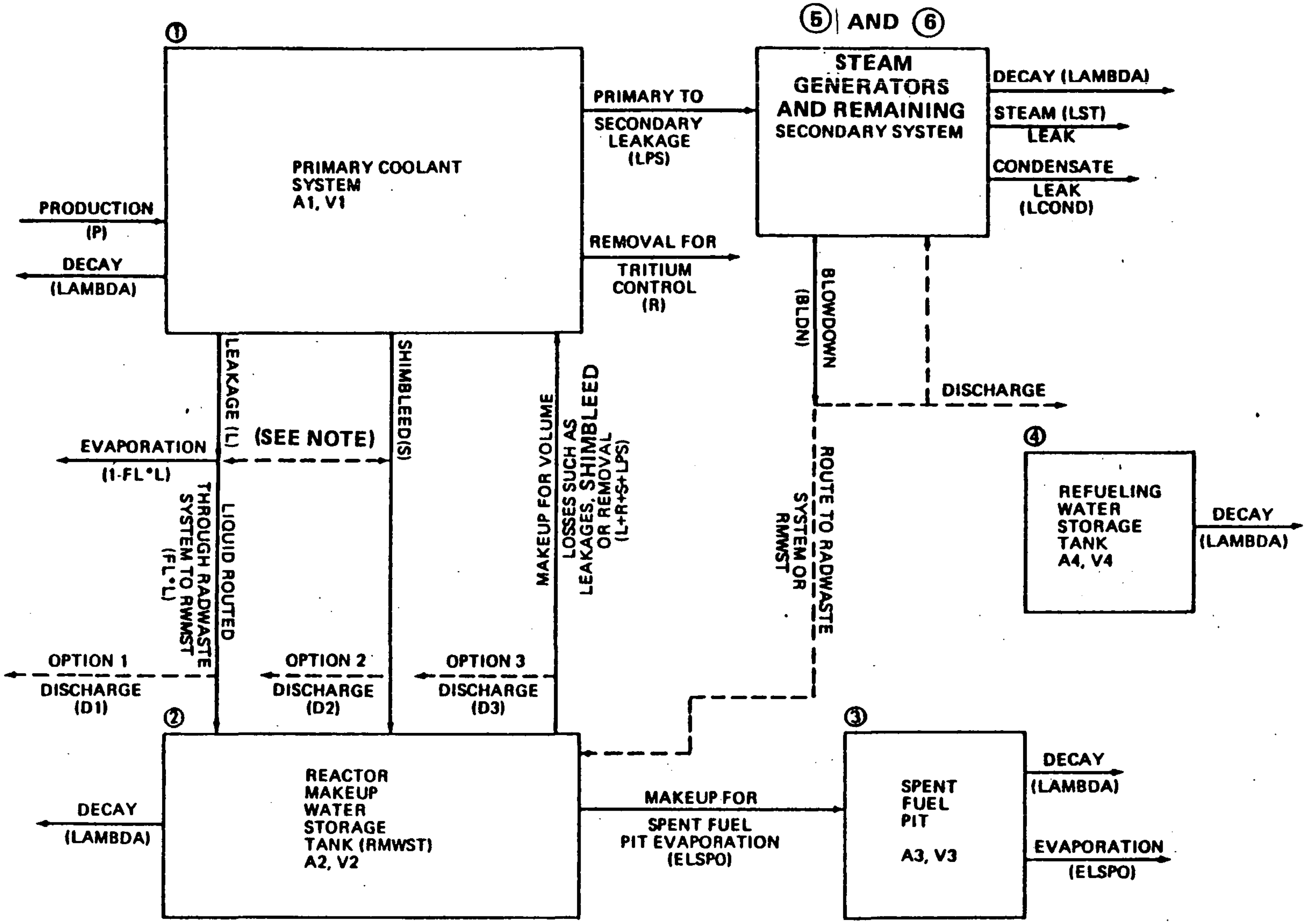


FIGURE 1 SCHEMATIC OF SYSTEMS

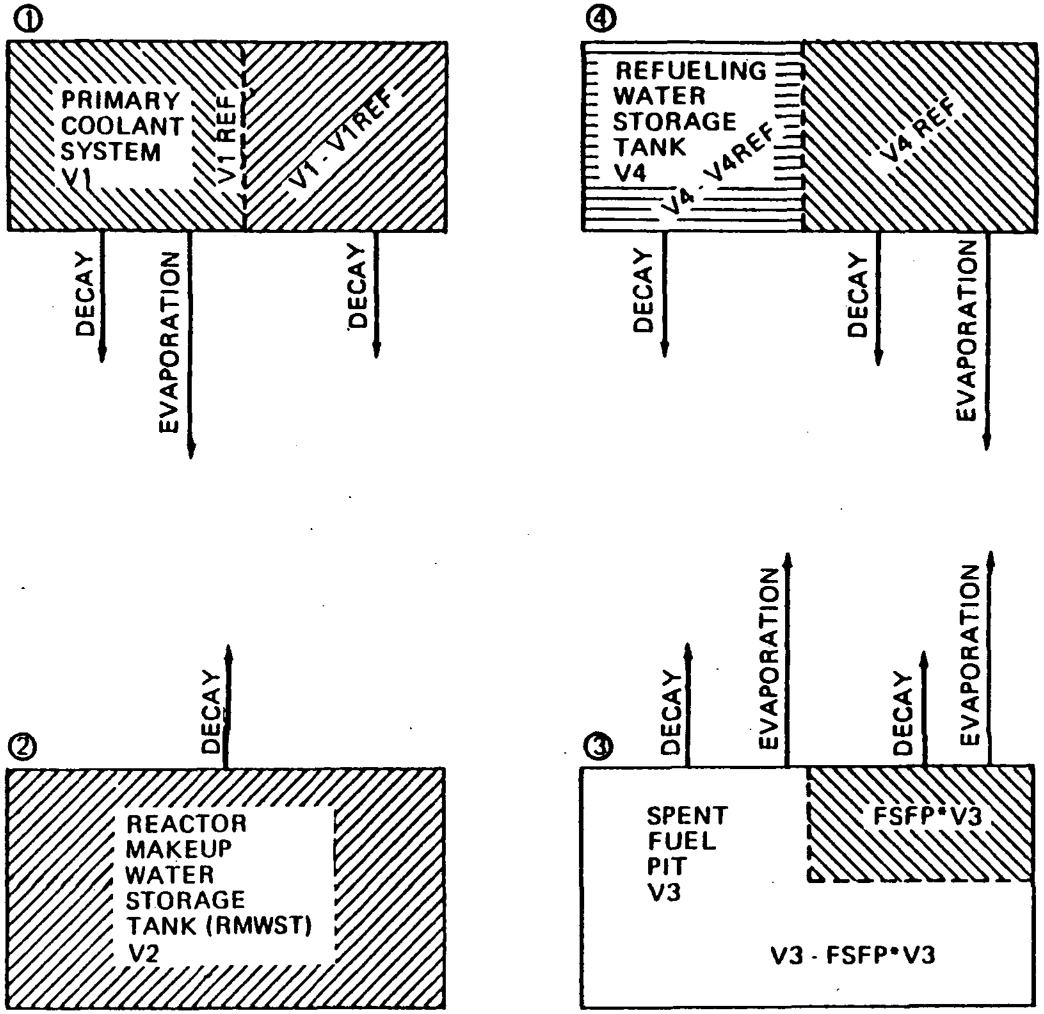
EQUIPMENT:

A. REACTOR & PRIMARY COOLANT SYSTEM
B. STEAM GENERATOR
C. LETDOWN DEMIN-ERALIZER
D. BORON THERMAL REGENERATION SYSTEM (WESTINGHOUSE DESIGN)
E. VOLUME CONTROL TANK
F. SHIM BLEED HOLDUP TANK
G. DEMINERALIZER(S)
H. BORIC ACID EVAPORATOR
I. REACTOR MAKEUP WATER STORAGE TANK (RMWST)
J. REACTOR COOLANT DRAIN TANK(S) (RCDT)
K. SPENT FUEL PIT
L. REFUELING WATER STORAGE TANK
M. TURBINE
N. CONDENSER
O. EVAPORATION
P. PRODUCTION, Ci/yr.
R. REMOVAL FOR TRITIUM CONTROL
S. SHIM BLEED
T. BLOWDOWN (NOTE: OPTIONS TO ROUTE: TO ENVIRONMENT; MAIN CONDENSER RADWASTE SYSTEM, OR RMWST
U. STEAM LEAK
V. CONDENSATE LEAK
W. PRIMARY TO SECONDARY LEAK
X. PRIMARY COOLANT LEAK
Y. RACTION OF PRIMARY COOLANT LEAK LEAVING AS A GAS
Z. RADIOACTIVE DECAY
AA. DISCHARGE FROM RCDT
BB. DISCHARGE OF SHIMBLEED
CC. DISCHARGE FROM RMWST



NOTE: IN MODELING THE LEAKAGE AND SHIMBLEED THIS ROUTING OPTION IS ELIMINATED SINCE A SIMILAR CONFIGURATION IS POSSIBLE VIA MANIPULATION OF DISCHARGE OPTIONS.

FIGURE 2 BLOCK DIAGRAM OF SYSTEMS DURING ROUTINE OPERATION



VOLUME IN PRIMARY COOLANT SYSTEM AND REFUELING CANAL DURING REFUELING

VOLUME STORED IN RMWST DURING REFUELING

VOLUME OF SPENT FUEL PIT WATER NOT MIXED WITH PRIMARY COOLANT

VOLUME REMAINING IN REFUELING WATER STORAGE TANK

<u>DISCHARGE FROM THE REACTOR DURING REFUELING</u>

① LAMBDA*A1, A2, A3, A4 (RADIOACTIVE DECAY)

② EVAPORATION FROM REFUELING CANAL (ELRCR) AND FROM SPENT FUEL PIT (ELSFPR)

FSFP FRACTION OF SPENT FUEL PIT WATER MIXING WITH PRIMARY COOLANT SYSTEM WATER

EVAPORATION IN SPENT FUEL PIT

$$\frac{30.4*ELSFPR*A3*(1\text{-}FSFP)}{V3\text{-}FSFP*V3}$$

EVAPORATION IN REFUELING CANAL

$$\frac{30.4*ELRCR*(A1*V1REF/V1 + A3*FSFP + A4*V4REF/V4)}{V1 + FSFP*V3 + V4REF}$$

FIGURE 3 REFUELING VOLUMES

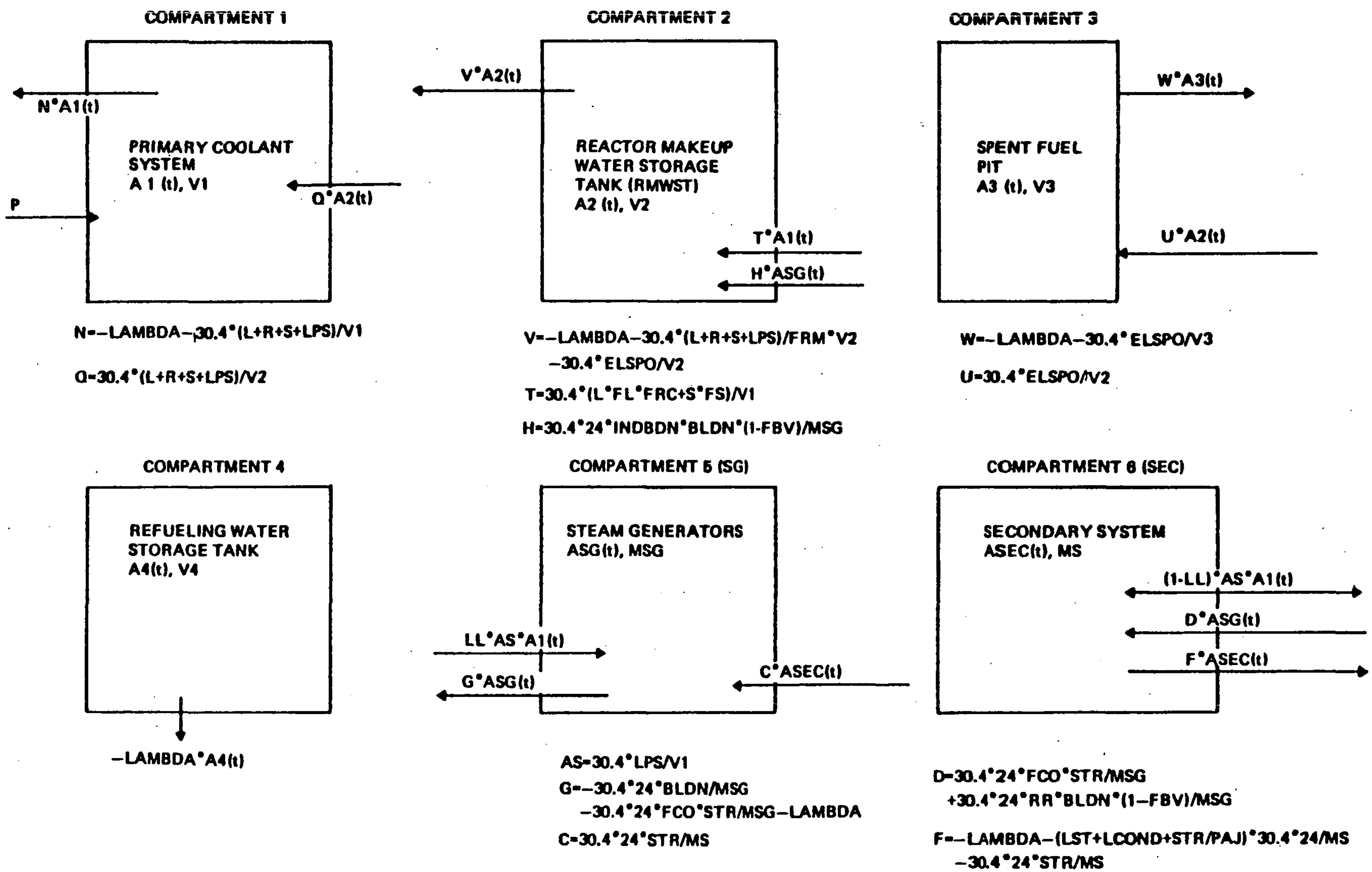


FIGURE 4 MATHEMATICAL MODELS OF THE COMPARTMENTS

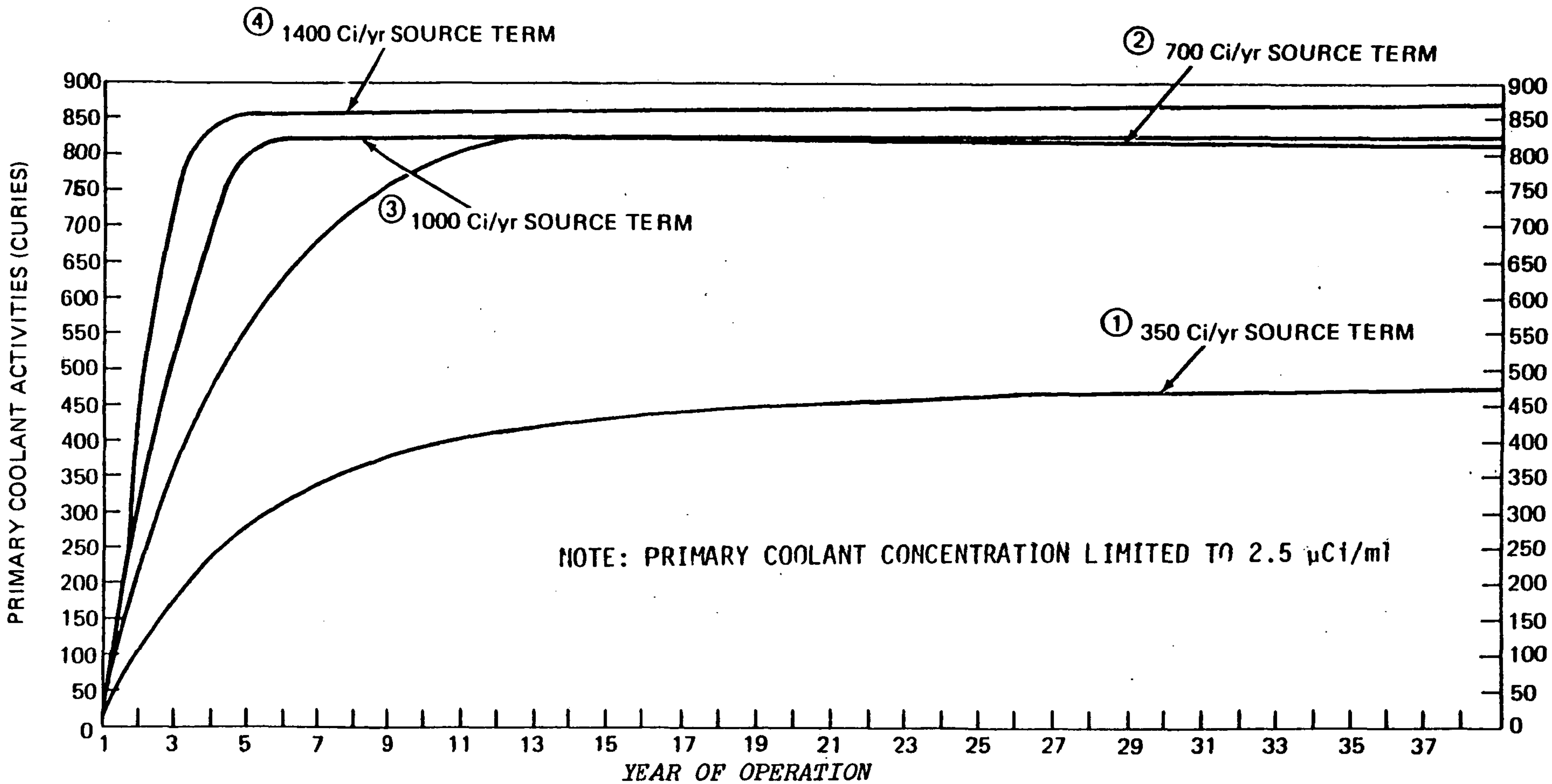


FIGURE 5 PRIMARY COOLANT ACTIVITIES (PRIOR TO REFUELING)

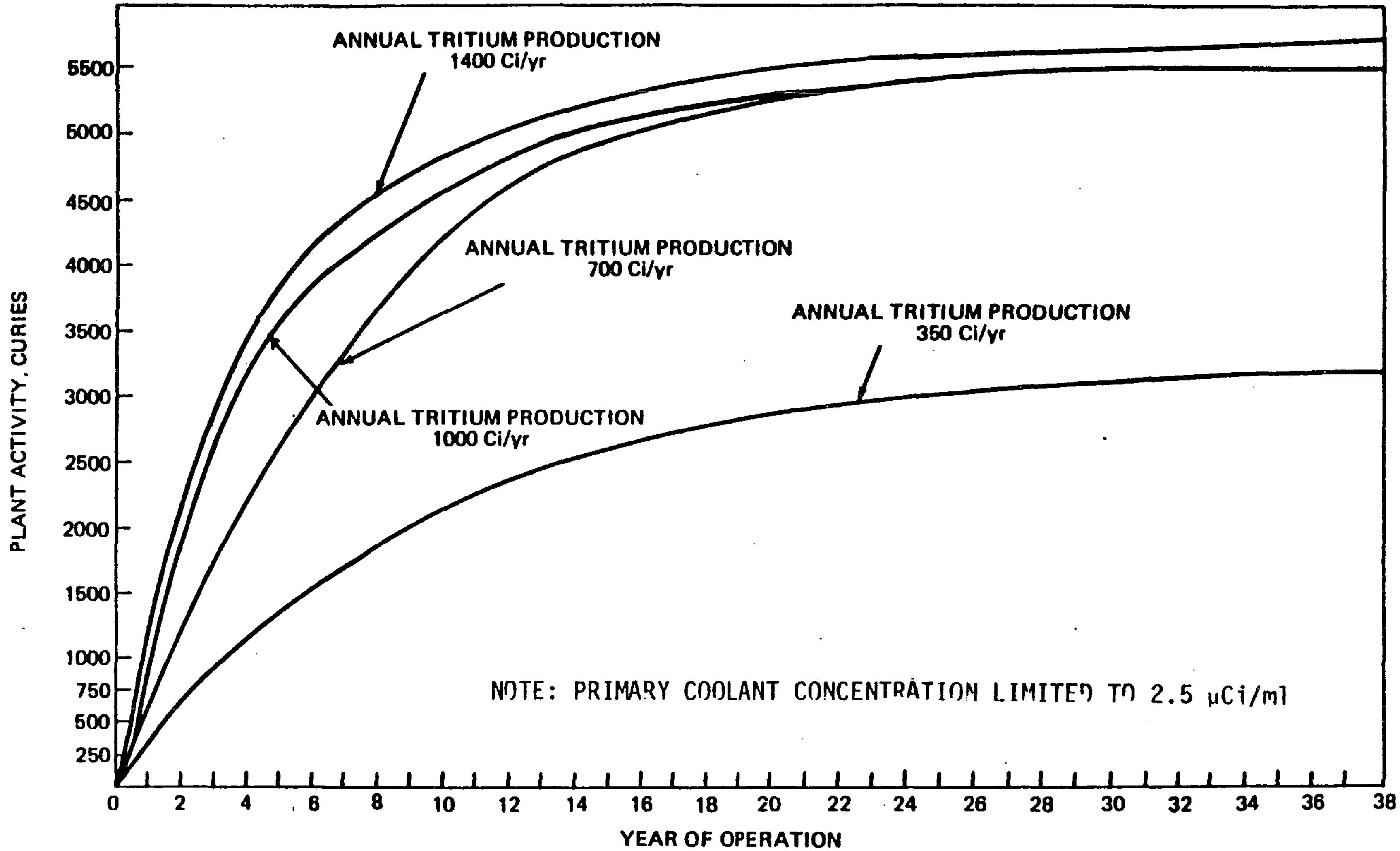


FIGURE 6 PLANT ACTIVITY VERSUS TIME

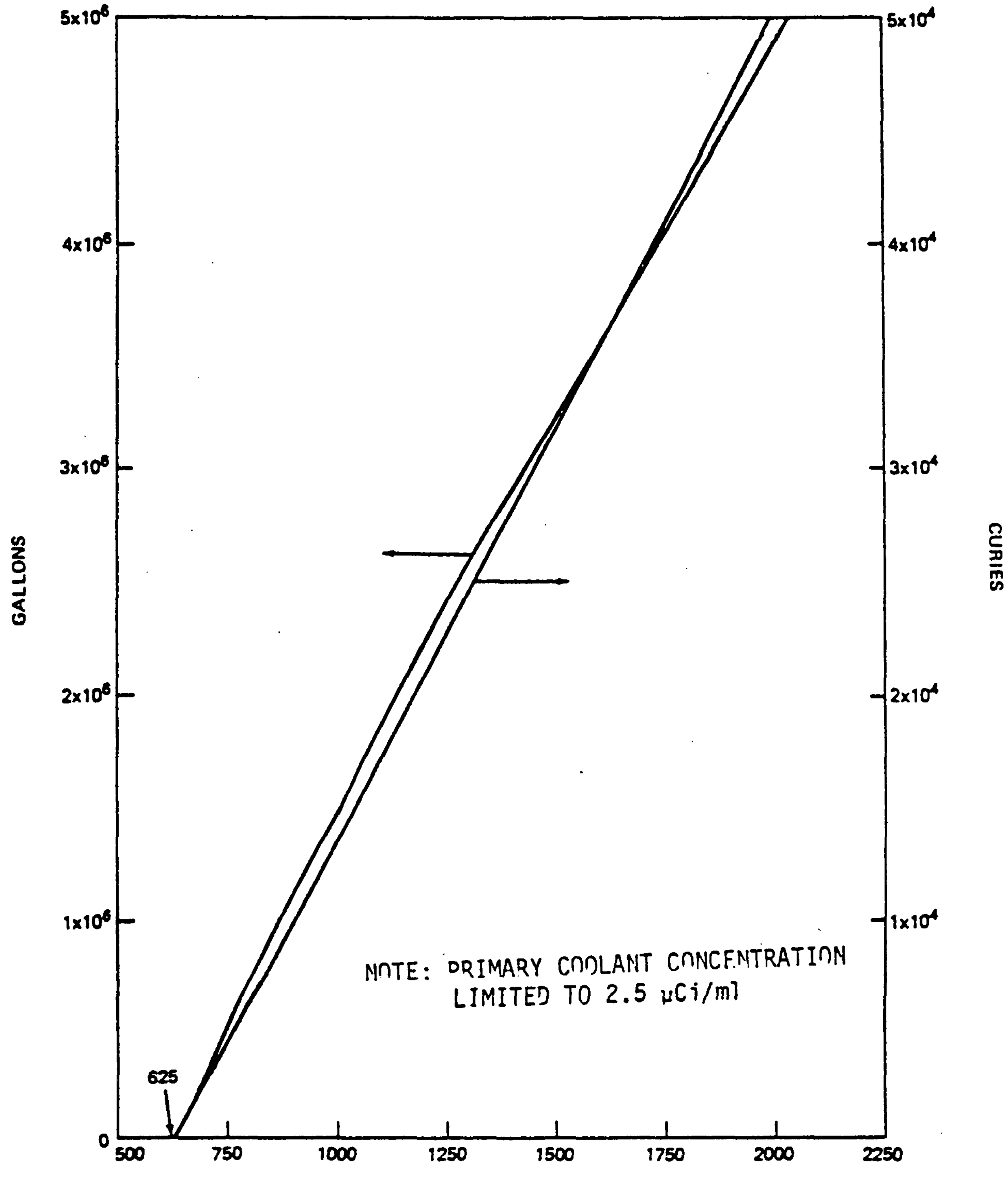


FIGURE 7 INTENTIONAL REMOVAL VS. SOURCE TERM

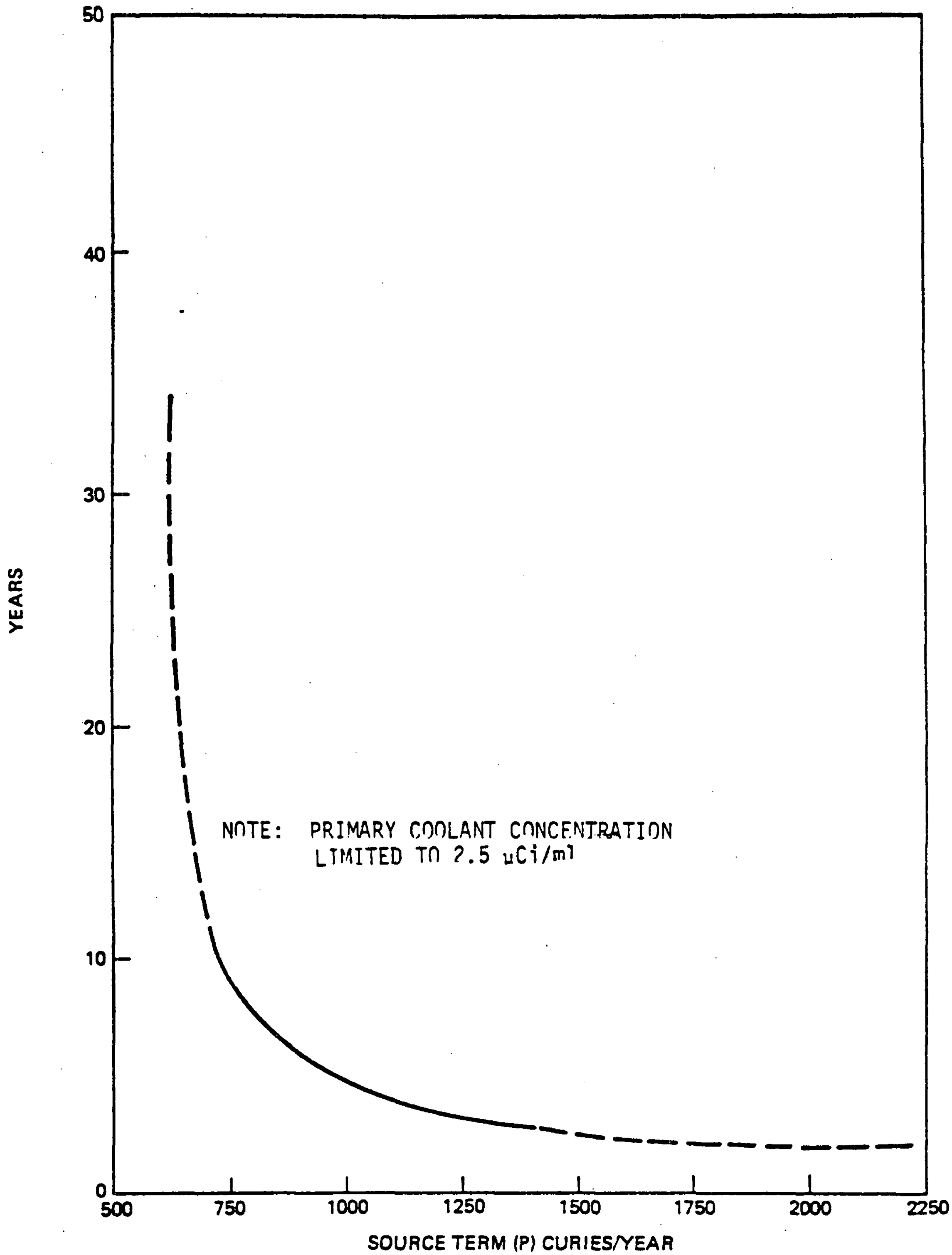
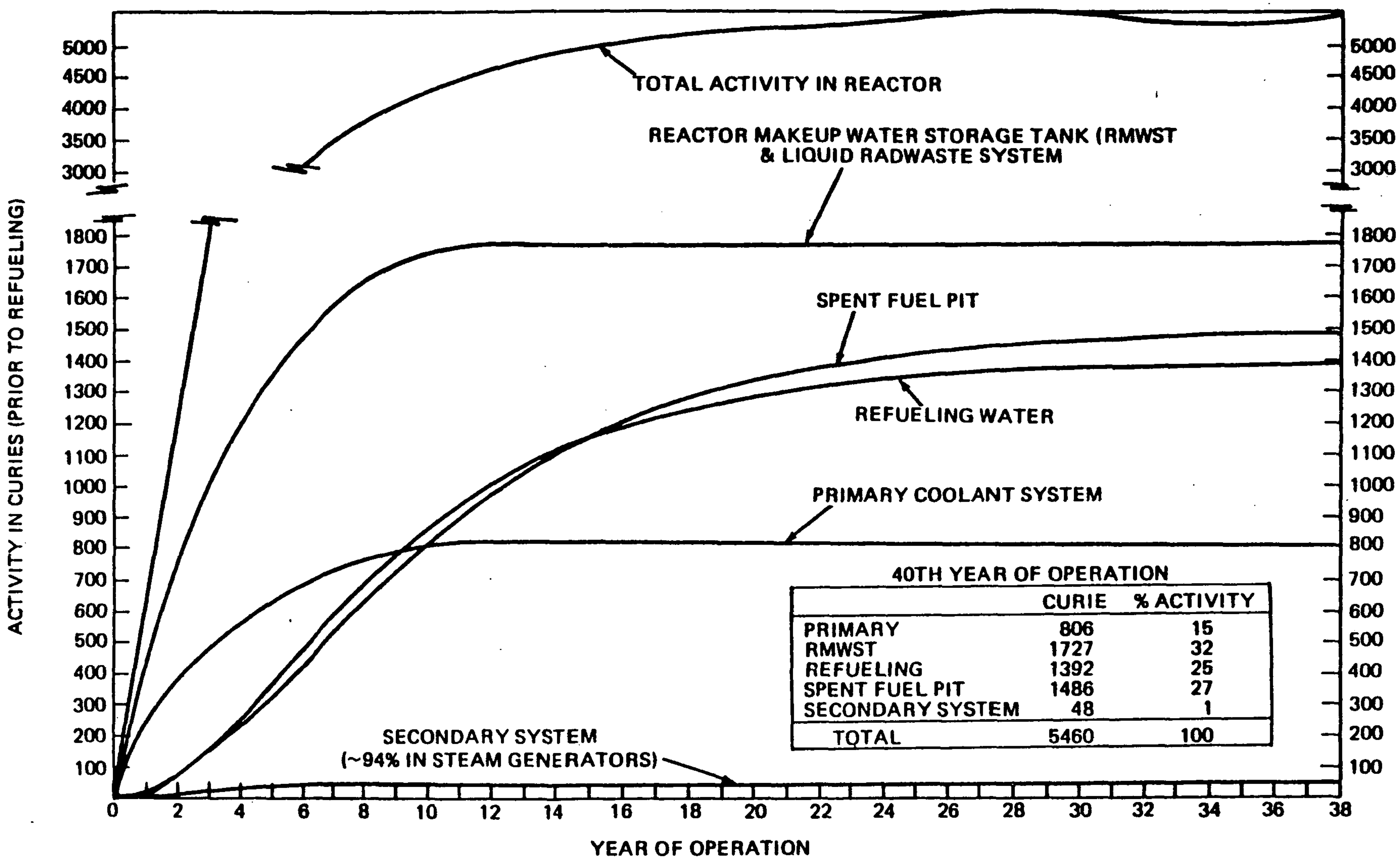


FIGURE 8 TIME UNTIL REMOVAL BEGINS VS. SOURCE TERM



	CURIE	% ACTIVITY
PRIMARY	806	15
RMWST	1727	32
REFUELING	1392	25
SPENT FUEL PIT	1486	27
SECONDARY SYSTEM	48	1
TOTAL	5460	100

FIGURE 9 ACTIVITY BUILDUP IN EACH COMPARTMENT (700 Ci/yr SOURCE TERM)

NOTE: PRIMARY COOLANT CONCENTRATION LIMITED TO 2.5 μCi/ml

BLOCK 1
ANNUAL AVERAGES OVER A 40 YEAR PERIOD

	Ci/yr	GAL/YR
DECAY	253	–
ENVIRONMENTAL	241	–
EVAPORATION	(101)	–
CONTAINMENT	(91)	–
SEC. SYSTEM	(49)	
REMOVAL	70*	7500*
RETENTION	137	–

BLOCK 2
40 YEAR TOTALS

	CURIES	GALLONS	FRACTION OF SOURCE TERM, %
DECAY	10,120	–	36
ENVIRONMENTAL	9,640	–	34
EVAPORATION	(4,000)		(14)
CONTAINMENT	(3,600)		(13)
SEC. SYSTEM	(2,000)		(7)
REMOVAL	2,780	300,000	10
RETENTION	5,460	1,044,000**	20

BLOCK 3
40TH YEAR OF OPERATION

	Ci/yr	%
DECAY	305	44
ENVIRONMENTAL		
EVAPORATION	136	19
CONTAINMENT	99	14
SEC. SYSTEM	55	8
REMOVAL	105	15
RETENTION	0	0

*DURING THE YEARS OF ACTUAL REMOVAL (12 THROUGH 40) THE ANNUAL AVERAGES ARE 100 Ci/yr AND 11,000 GAL/DAY AND THE CONCENTRATION IS 2.46 μ Ci/ml

**EXCLUDES SECONDARY SYSTEM VOLUMES WHICH CONTAIN VERY LITTLE TRITIUM

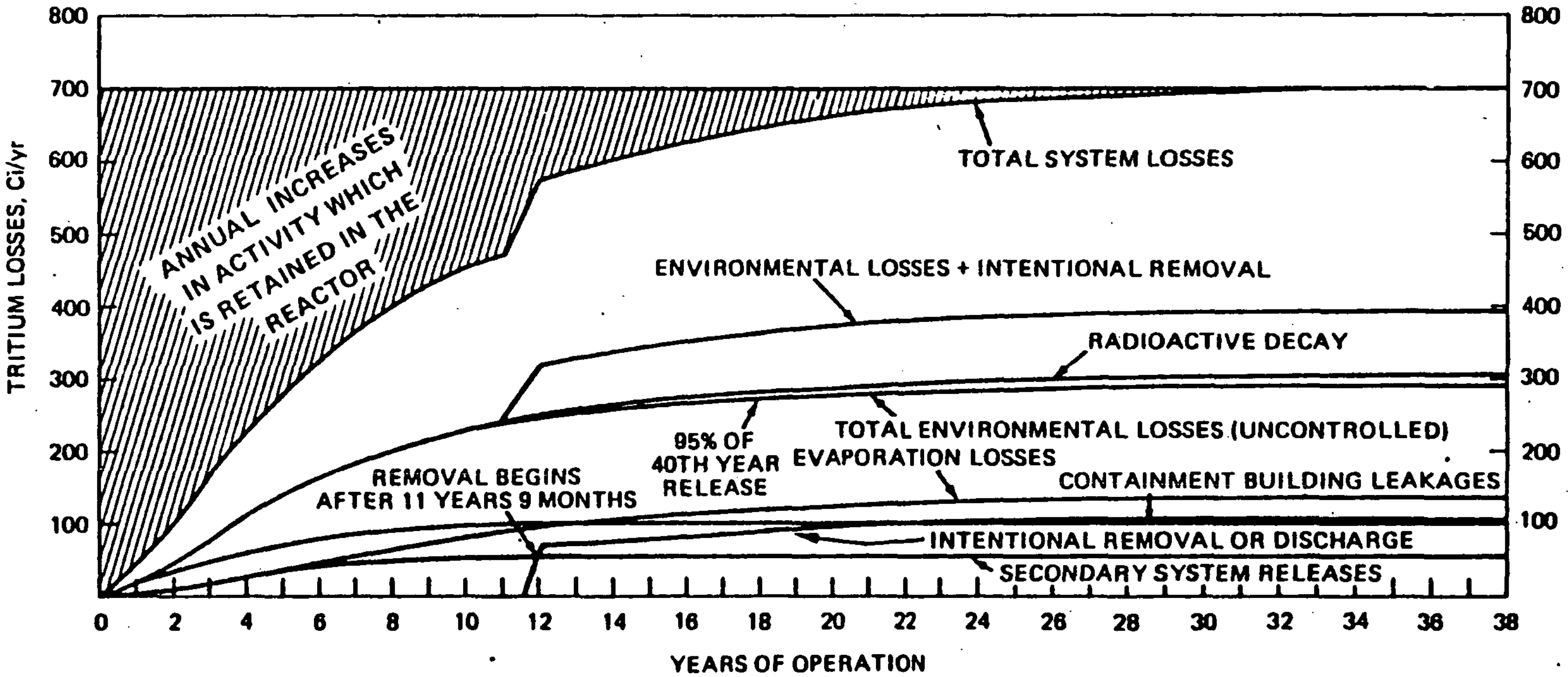


FIGURE 10 TRITIUM LOSSES FROM BASE PLANT DESIGN (700 Ci/yr SOURCE TERM)
NOTE: PRIMARY COOLANT CONCENTRATION LIMITED TO 2.5 μ Ci/ml

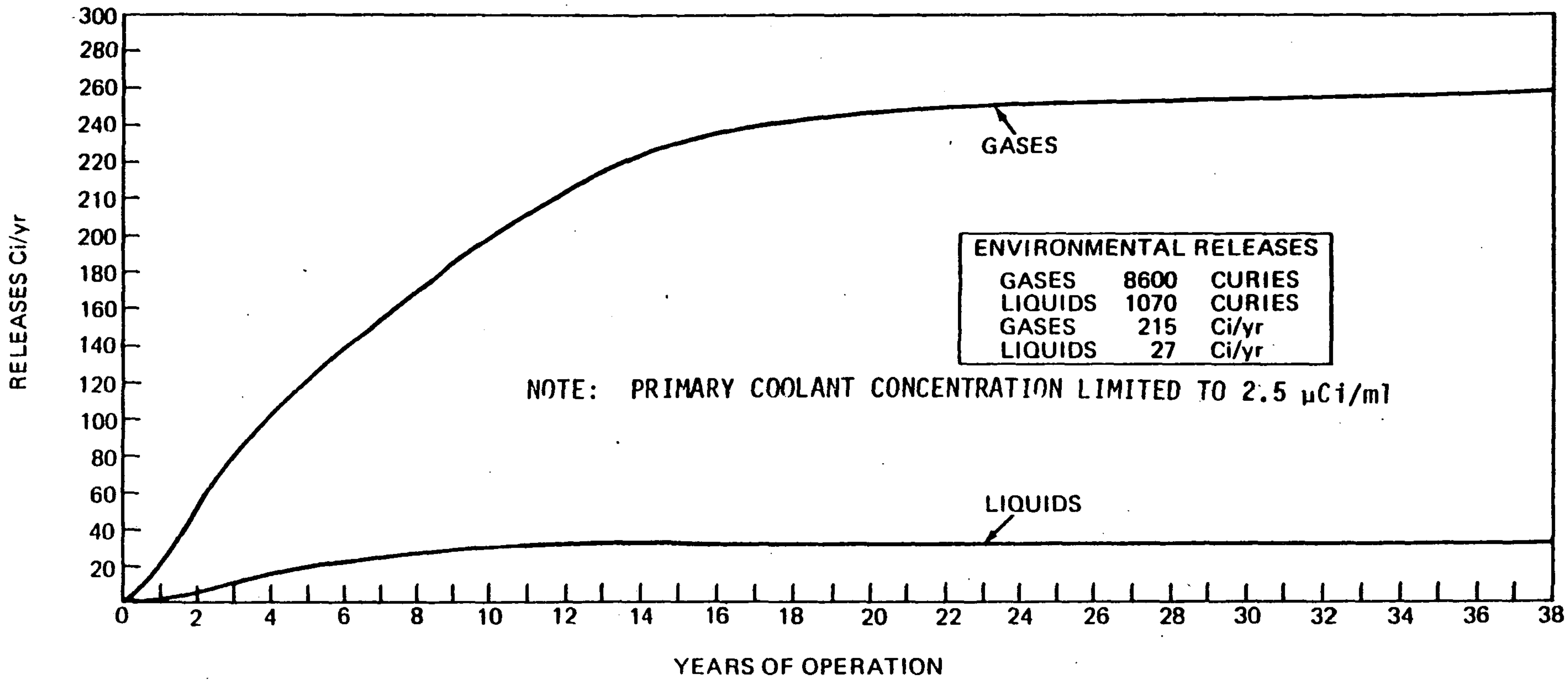


FIGURE 11 LIQUID AND GASEOUS ENVIRONMENTAL RELEASES (700 Ci/yr SOURCE TERM)

0.0 BLOWDOWN RECYCLE TO MAIN CONDENSER	8400 LB/HR BLOWDOWN DIRECT RELEASE TO THE ENVIRONMENT	84,000 LB/HR BLOWDOWN DIRECT RELEASE TO THE ENVIRONMENT
40 YEAR RELEASES (TOTAL PLANT)	40 YEAR RELEASES (TOTAL PLANT)	40 YEAR RELEASES (TOTAL PLANT)
LIQUIDS = 1070 CURIES	LIQUIDS = 1996. CURIES	LIQUIDS = 2000. CURIES
GASES = 8600 CURIES	GASES = 7911 CURIES	GASES = 7907. CURIES
TOTAL = 9670 CURIES	TOTAL = 9907 CURIES	TOTAL = 9907. CURIES

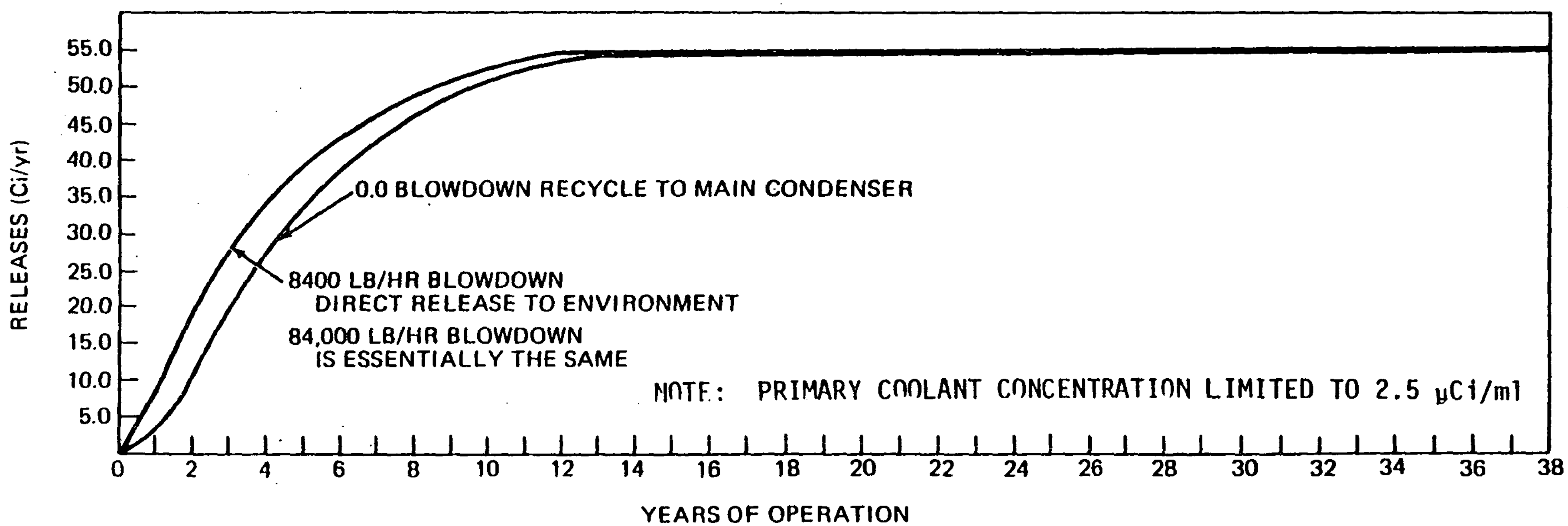


FIGURE 12 SECONDARY SYSTEM RELEASES VERSUS STEAM GENERATOR BLOWDOWN RATE (700 Ci/yr SOURCE TERM)

40 YEAR TOTALS

	0.0 LEAK RATE	18.7 gpm LEAK RATE
LIQUID RELEASES	0.0 CURIES	1070 CURIES
GASEOUS RELEASES	7800 CURIES	8600 CURIES
TOTAL ENVIRONMENTAL RELEASES	7800 CURIES	9670 CURIES
INTENTIONAL REMOVAL	4480 CURIES	2780 CURIES
RADIOACTIVE DECAY	10,300 CURIES	10,120 CURIES
ACTIVITY RETAINED	5400 CURIES	5460 CURIES
REMOVAL (LIQUID VOLUME)	500,000 GALLONS	300,000 GALLONS
REMOVAL BECAME NECESSARY AT	8 YEARS, 11 MONTHS	11 YEARS, 9 MONTHS

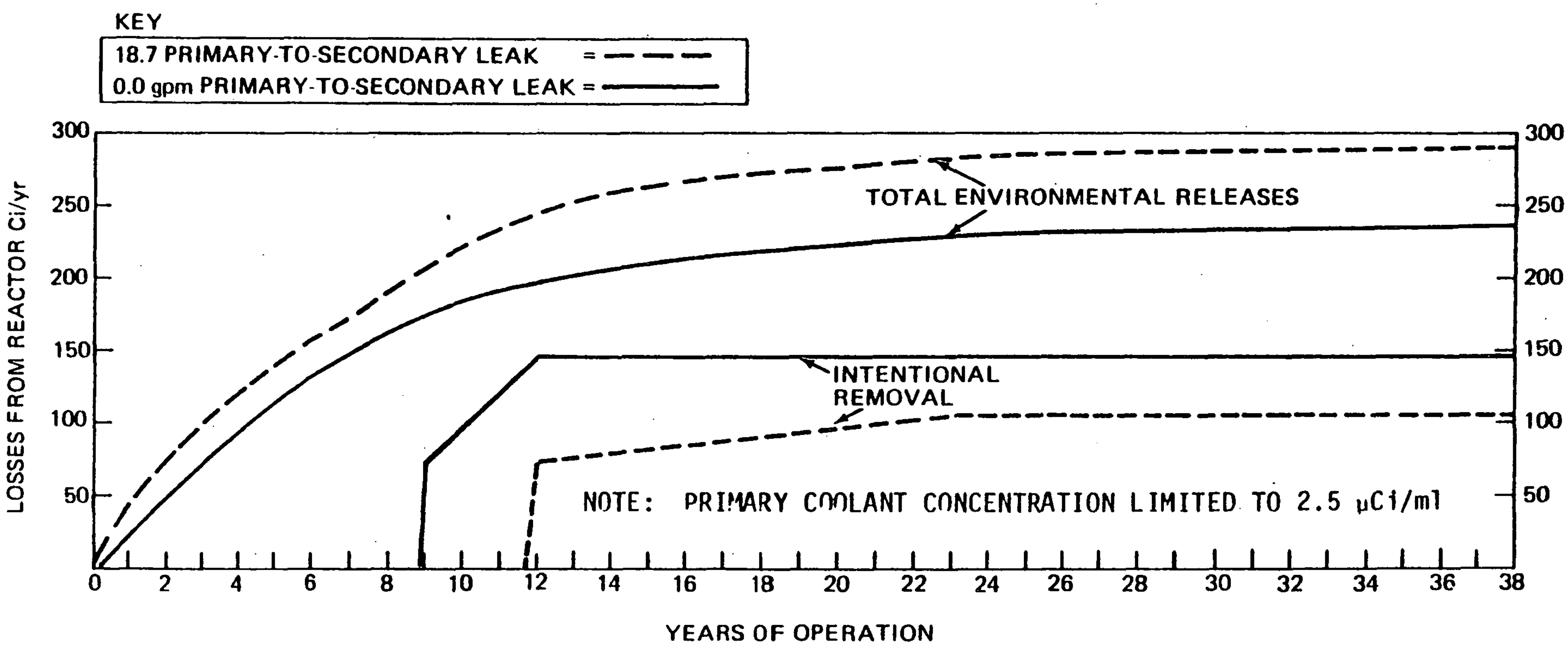


FIGURE 13 EFFECT OF PRIMARY-TO-SECONDARY LEAKAGE (700 Ci/yr SOURCE TERM)